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Comparison of H-Mode Plasmas Diverted to Solid and Liquid Lithium Surfaces

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Abstract

Experiments were conducted with a Liquid Lithium Divertor (LLD) in NSTX. Among the goals was to use lithium recoating to sustain deuterium (D) retention by a static liquid lithium surface, approximating the ability of flowing liquid lithium to maintain chemical reactivity. Lithium evaporators were used to deposit lithium on the LLD surface. Improvements in plasma edge conditions were similar to those with lithiated graphite plasma-facing components (PFCs), including an increase in confinement over discharges without lithium-coated PFCs and ELM reduction during H-modes. With the outer strike point on the LLD, the D retention in the LLD was about the same as that for solid lithium coatings on graphite, or about two times that achieved without lithium PFC coatings. There were also indications of contamination of the LLD surface, possibly due erosion and redeposition of carbon from PFCs. Flowing lithium may thus be needed for chemically active PFCs during long-pulse operation.

PSI19 Keywords: NSTX, Lithium, Recycling, Coating, Divertor material

JNM Keywords: C0600 Coatings; P0500 Plasma-materials interaction; S1300 Surface effects; I0100 Impurities.

PAC Numbers: 52.25.Vy; 52.40.-w; 52.40.Hf; 52.55.Fa.

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1. Introduction

Reduction in edge recycling in spherical torus divertor plasmas has been observed with solid lithium PFCs [1]. This occurs through the formation of lithium compounds or complexes that bind deuterium [2]. Solid lithium coatings eventually saturate, however, and thus provide only short-pulse capability. Liquid lithium contained in a suitable medium has a much higher capacity for absorbing D, and in addition has potential to provide self-healing walls for high power reactors. Improvements in plasma performance in the National Spherical Torus Experiments (NSTX), using solid lithium coatings on a graphite divertor surface, motivated the installation of a fully-toroidal Liquid Lithium Divertor (LLD) [3-5].

In 2010, NSTX experiments were conducted using the LLD, which was installed on the large major radius side of the lower divertor [5]. Figure 1 shows the LLD as it appeared prior to the start of plasma operations. The LLD surface consisted of a 0.165 mm thick layer of plasma-sprayed molybdenum, with a porosity of 45% (Fig. 2). Lithium evaporators (LITERs) were used to deposit lithium on the LLD surface [1]. The purpose of the LLD was to test the effectiveness of maintaining the D retention properties of a continuously replenished liquid lithium surface. For the LLD, this was effected by lithium evaporation onto its surface instead of flowing liquid lithium.

The LLD consists of four 22 cm wide plates, each having a toroidal extent of 82.5°. The quadrants are separated toroidally by graphite tiles, which contain diagnostics and electrodes for edge plasma biasing [6]. The LLD plates are mode of 2.2 cm thick copper with a 0.25 mm stainless steel liner bonded to its surface. This provides a barrier between the copper and the lithium in the porous molybdenum plasma-facing surface. The porosity of the molybdenum facilitates the wetting and subsequent spreading of liquid lithium over the LLD. It also insures that surface tension forces are large relative to electromagnetic forces to retain the liquid lithium during plasma operations [7].

The lithium capacity of the porous LLD surface was 37 g. The 2010 experiments were performed with the LLD filled with increasing amounts of lithium, from a few percent up to

100% of capacity, over the duration of the campaign. Each of the four LLD plates contained embedded electrical resistive heaters and thermocouples to monitor the heating. Two LITERs in the upper dome of the NSTX vacuum vessel were able to evaporate lithium over the entire LLD surface. They were used to deposit, over 10-minute intervals, between 100 and 700 mg of lithium in about 90% of NSTX discharges.

Figure 3 shows an elevation of NSTX with the location of the two LITERs and the LLD. Each LITER consists of a reservoir that is heated to liquefy the lithium. The lithium vapor enters the NSTX vacuum chamber through an exit tube that is aimed toward the lower divertor region, and covers the area where the LLD is located. As with litiated graphite, plasmas with the LLD as a plasma-facing component (PFC) demonstrated improvements over PFCs without lithium coatings that included higher confinement and ELM reduction during H-modes.

2. Experimental Results

During the 2010 NSTX campaign, experiments were performed with the outer strike point on the LLD (Fig. 4). Similar D fueling rates were needed to maintain the operating conditions achieved with solid lithium coatings on graphite prior the installation of the LLD. This was about twice as high as level required when lithium coatings were not present, and suggests a factor of two increase in D retention rate that did not change in the presence of the LLD.

In discharges with the outer strike point (OSP) on the LLD, confinement improved over plasmas without lithium-coated PFCs. Additional features reminiscent of plasma behavior with lithiated graphite included ELM suppression and volume-average plasma lithium concentrations <0.1%. Two-color measurements with IR camera showed no hot spots, indicating that the thermal response of the LLD during discharges was determined by the effective heat distribution by the copper substrate.

At 500 to 600 ms into the discharge, the measured LLD surface ranged from

160 to 300°C in the vicinity of the OSP. The plasma at the outer strike point was thus incident on liquid lithium in many discharges. There was no indication of macroscopic lithium ejection from the LLD, however, as observed in other experiments where lithium was exposed to high-power divertor plasmas [8].

The absence of lithium ejection is also supported by the lack of any indication that either the molybdenum PFC or stainless steel liner for the LLD was exposed to the plasma. There was little spectroscopic evidence for radiation from metallic impurities, and examination of the plasma-sprayed molybdenum or the underlying liner after the run campaign did not reveal any macroscopic damage. However, there were spectroscopic indications of increasing deuterium desorption during discharges with OSP on the LLD lithium surface. Evidence for carbon and oxygen impurity contamination was also observed. This suggests that while the LLD is able to retain bulk lithium during high-power divertor discharges, its surface behavior is complex.

To further investigate the characteristics of the LLD as a PFC, two experiments were performed with two different approaches for varying the LLD temperature. In the first experiment, the same fueling scenario, i. e., gas puff programming, was maintained as the LLD temperature was varied. This was accomplished with heaters that were embedded in each of the LLD plates. As the LLD surface temperature transitioned through the lithium melting point, the core plasma electron density and deuteron particle content remained relatively constant (Fig. 5). This indicated that D absorption at the solid and liquid lithium temperatures was the same. During this sequence, however, the plasma C^{6+} content decreased as the LLD transitioned through the lithium melting point (Fig. 6). The origin of this behavior is not understood, but was coincident with increasing ELMs. Performing the reverse experiment by proceeding from liquid to solid may facilitate an understanding of this behavior.

In the second experiment, the plasma itself was used to increase the LLD temperature. The LLD was heated by the discharges alone over a series of shots, with bulk temperature increasing at a rate of ~10°C/shot. As the LLD temperature rose, an apparent change in fueling efficiency was observed. It became necessary to increase the D fueling significantly to maintain plasma density and stability.

In Fig. 7, the total number of D particles puffed (fueling), and the resultant D core particle content, and C^{6+} core particle content are plotted as a function of the LLD surface temperature during the plasma. Once the density was recovered, the D fueling was deliberately extended to significantly high values of deuterium gas input. The core D density remained relatively unchanged, however, and the plasmas remained stable.

3. Discussion

The similarity of results obtained with the LLD temperature controlled through embedded heaters and solid lithiated graphite may be due to the fact that in each case, the behavior of the PFC is determined by the lithium coating on a chemically-inert substrate. Although the temperature of the LLD exceeded the melting point of lithium, it never rose about the temperature of the lithium compound with the lowest melting point, i. e., lithium hydroxide at 462°C. A lithium compound layer can thus be present on the LLD surface, and this possibility has been corroborated with laboratory tests where an LLD sample was exposed to a high-power neutral beam.[9] Power densities approaching levels found in NSTX were reached, but the lithium compound layer on persisted on the surface facing the beam.

If there is a difference in the plasma behavior when the OSP was on the LLD, it may not be due to the interaction of more energetic incident ions with the LLD surface. The argument for the existence and persistence of a lithium compound layer when embedded heaters were used also apply for LLD heating with plasmas, where the maximum surface temperature was in the vicinity of 300°C. To estimate the range of incident D particles in the lithium compound layer, data from a 99-probe High-Density Langmuir Probe (HDLP) array were used. The HDLP array was installed between the LLD plates (Fig. 1). Assuming that the ion temperature was equal to the electron temperature, an incident D energy of about 50 eV was deduced at the OSP. This result is consistent with midplane Charge Exchange Recombination Spectroscopy (CHERS) measurements.

Fig. 8 shows the results of a calculation with the TRIM code [10]. They indicates a stopping range of about 5 nm in lithium for 50 eV deuterium ions, and a stopping range of about 3 nm in typical lithium compounds formed with common residual gases in the vacuum environment. V_p is the typical floating potential of a Langmuir probe at the divertor strike point, and T_i the typical ion temperature in the plasma scrape-off. In addition to assuming that the ion temperature is equal to the electron temperature, the incident ion energy is taken as the sum of the measured plasma potential (V_p) and an assumed ion temperature contribution to the incident energy of twice the ion temperature (T_i). Also indicated is a deuterium ion energy estimate based on Kocan et al. [11] who report that T_i in various fusion devices could range from one to ten times the electron temperature in the scrapeoff layer (SOL).

In the case of 50 eV D, the stopping range in lithium is 5 nm. The range in typical lithium impurity compounds is reduced to about 3 nm. Figure 8 shows that even if the ion temperature was 10 times greater than the electron temperature [12], the stopping range is only about 6 nm. Impurity layers of this thickness were easily accrued during the 2010 experimental campaign. Static liquid lithium on the LLD getters the NSTX residual vacuum impurity gases H_2O, CO, and CO_2. X-ray photelectron spectroscopy (XPS) measurements at $5x10^{-8}$ torr find that O_2 and H_2O at 10^{-6} torr could oxidize 20 monolayers of lithium (5 nm) in 20 seconds. Under both LLD heating scenarios, this would result in a continuous codeposition of such species during the evaporation of lithium between shots.

Surface deposition is also expected to occur as plasmas interact with PFCs, including redeposition of eroded graphite PFCs in NSTX would also be codeposited on the lithium coatings. The dissimilarities in the D efflux during the two LLD heating experiments could lead to differences in the LLD surfaces during discharges, and hence affect how the D is retained. This is very much an open question, however, and laboratory work is in progress measure the relative retentions of D in lithium impurity complexes under a variety of conditions that simulate the tokamak environment.[13]

4. Conclusions

The NSTX LLD was implemented to provide a fully-toroidal lithium PFC for high power divertor plasmas. With the OSP on the LLD, its plasma-sprayed surface retained the lithium, and the thin stainless steel liner was an effective lithium barrier for the copper heat sink. Discharges with the LLD in place exhibited characteristics that were similar to plasmas with lithiated graphite PFCs, including improved confinement and ELM suppression. The degree to which these benefits were directly attributable to the LLD, however, is not clear. The experimental results, for example, could be explained if lithium was evaporated onto a lithium compound layer, rather than forming an actual liquid lithium LLD surface.

The fact that lithium evaporation was also the means for loading the LLD creates an additional complication. The total lithium deposition during the 2010 NSTX campaign was substantial, and amounted to 1.3 kg. The lithium distribution is basically Gaussian around the axis of the LITER exit tube, which means that more lithium in the "wings" of the distribution means more PFC coverage. Furthermore, the spatial extent of the lithium coatings on the PFCs suggests plasma-related transport effects that go beyond simple line-of-sight deposition. For this reason, the beneficial effects of the lithium in the 2010 experiments could be due primarily to increased lithium coverage of the graphite PFCs.

A flowing lithium LLD is thus needed to demonstrate the efficacy of a lithium PFC specifically in its liquid state. This approach is required to avoid the issues introduced by complications such as lithium compound layer formation and lithium loading through evaporation. Work has begun at PPPL to address the technological challenges that must be overcome for the implementation of a flowing LLD on the NSTX-Upgrade. The experience in developing a static LLD has been a key element in this process.

Acknowledgments

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References

[1] H.W. Kugel *et al.,* Physics of Plasmas **15** (2008) 056118.

[2] C.N. Taylor *et al.,* J. Appl. Phys. **109** (2011) 053306.

[3] H.W. Kugel *et al.,* Fusion Eng. Des. **84** (2009) 1125.

[4] R.E. Nygren *et al.*, Fusion Eng. Des. **84** (2009) 1438.

[5] H.W. Kugel, *et al.,* Fusion Eng. Des. (2011), doi:10.1016/j.fusengdes.2011.07010.

[6] S. Zweben *et al.,* J. Nuc. Mater. **390-391** (2009) 417.

[7] M. Jaworski *et al.,* accepted for presentation at 24th IAEA Fusion Energy Conference (2012).

[8] D.G. Whyte *et al.,* Fus. Eng. Des. **72** (2003) 133.

[9] T. Abrams *et al.,* this conference.

[10] R. Bastaz and J. A. Whaley, Fusion Eng. Des. **72** (2004) 111.

[11] M. Kocan *et al.*, J. Nuc. Mater. **415** (2011) S1133.

[12] J.F. Ziegler and J.M. Manoyan, Nucl. Instrum. Methods **B35** (1989) 215.

[13] C.H. Skinner *et al.,* this conference

Figure Captions

Fig.1 Photo of the interior of NSTX before the start of the 2010 experimental campaign. The Liquid Lithium Divertor (LLD) plates form the light-colored toroidal ring in the lower divertor region. A graphite tile containing diagnostic sensors is also indicated.

Fig. 2 Photo of the plasma-facing side of an LLD plate showing the 0.165 mm thick molybdenum surface with 45% porosity. The molybdenum was plasma-sprayed onto a 0.25 mm thick stainless steel liner that was brazed to a 22.2 mm thick copper substrate.

Fig. 3 Schematic cross-section of NSTX showing location of the two Lithium Evaporators (LITERs) on the upper dome and the LLD on the lower dome. The shaded regions depict the 1/e widths of the emitted Gaussian-like distribution.

Fig. 4 Equilibrium reconstruction for plasma shape used in LLD experiments. The low-elongation discharges have the outer strike point incident on the LLD and the inner strike point incident on lithiated graphite inboard of the LLD.

Fig. 5 Core D particle content determined with Charge Exchange Recombination Spectroscopy (CHERS) and the volume average plasma electron content measured using Muliti-Point Thompson Scattering (MPTS) as a function of LLD surface temperature.

Fig. 6 Plasma C^{6+} content as function of LLD surface temperature. Carbon level decreased as the LLD transitioned from solid to liquid lithium temperatures.

Fig. 7 Fueling scan data at 0.5s for discharges with outer strike point on LLD. Shown is total number of deuterium particles puffed (fueling), resultant D core particle content, and C^{6+} core particle content as a function of LLD surface temperature.

Fig. 8 Results from TRIM code for various PFC materials and lithium compounds. The calculations indicate a stopping range of about 5 nm in lithium for 50 eV deuterium ions, and a

stopping range of about 3 nm in typical lithium compounds formed with common residual gases in the vacuum environment. V_p is the typical floating potential of a Langmuir probe at the divertor strike point and T_i the typical ion temperature in the plasma scrape-off region.

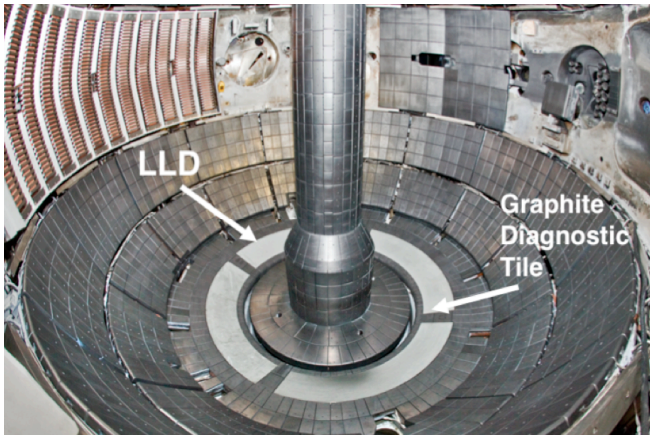

Fig. 1

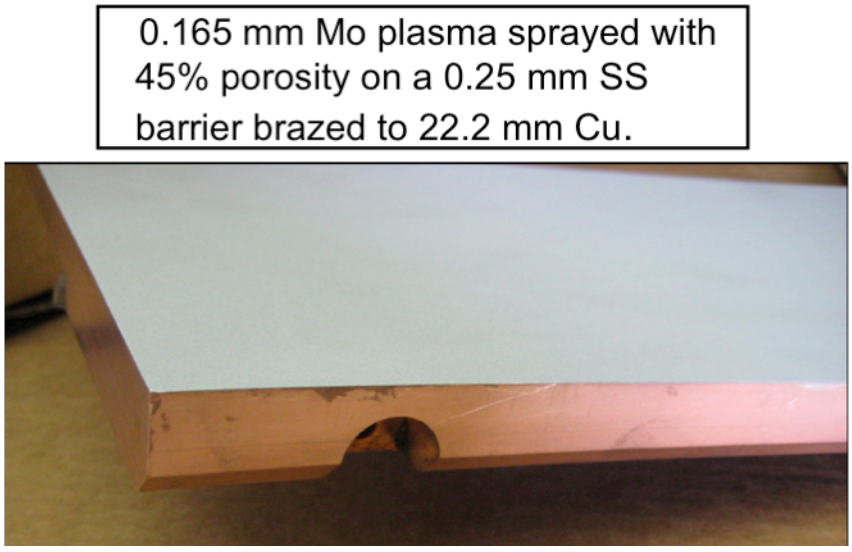

Fig. 2

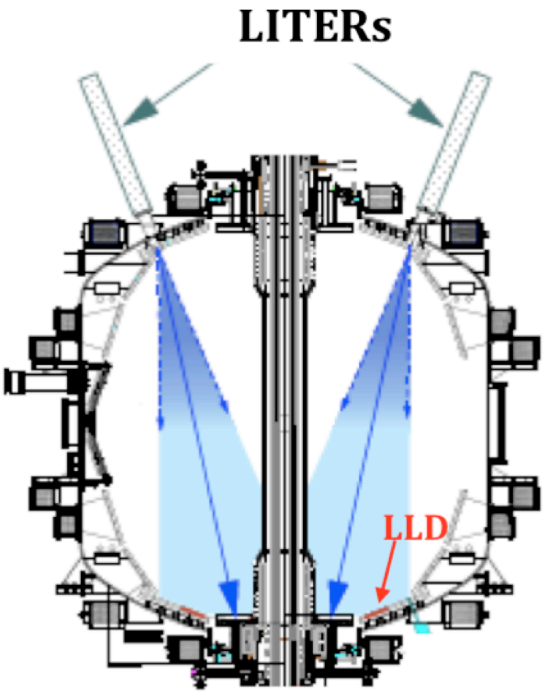

Fig. 3

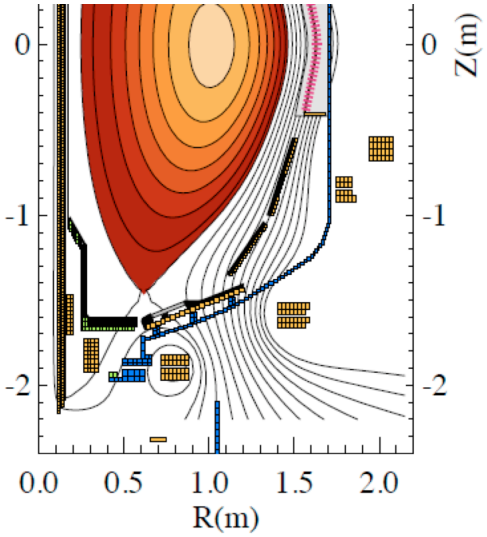

Fig. 4

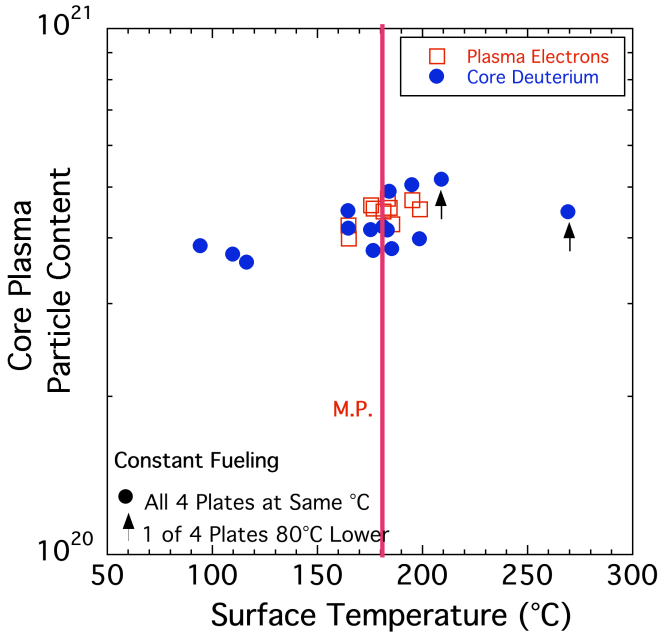
Plasma Electrons
Core Deuterium
Core Plasma
Particle Content
M.P.
Constant Fueling
All 4 Plates at Same °C
1 of 4 Plates 80°C Lower
Surface Temperature (°C)

Fig. 5

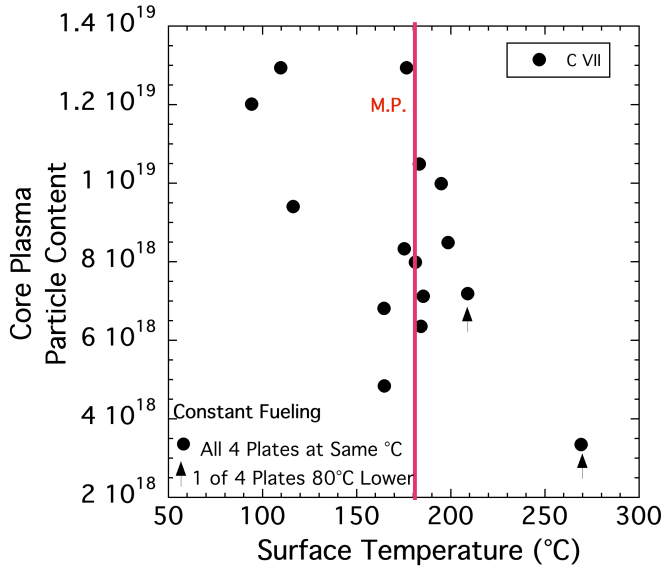
C VII
M.P.
Core Plasma
Particle Content
Constant Fueling
All 4 Plates at Same °C
1 of 4 Plates 80°C Lower
Surface Temperature (°C)

Fig. 6

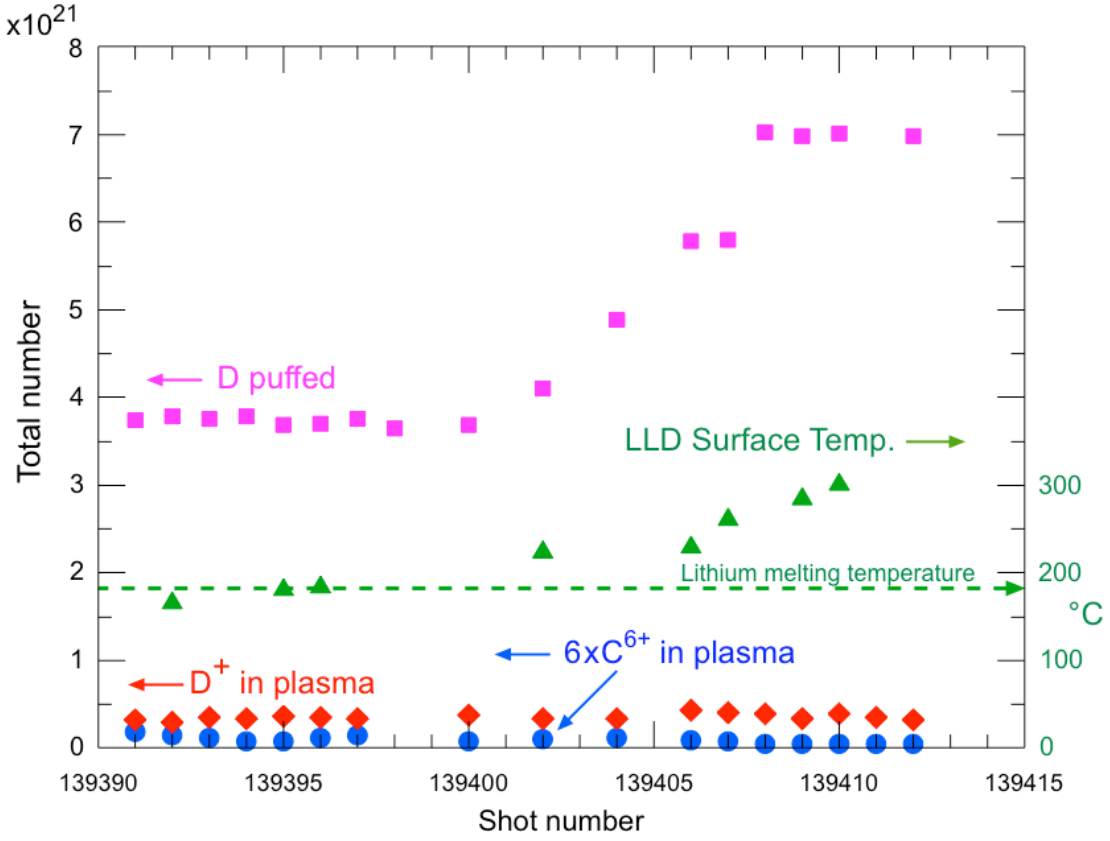
x10^21
Total number
D puffed
LLD Surface Temp.
Lithium melting temperature
6xC6+ in plasma
D+ in plasma
°C
Shot number

Fig. 7

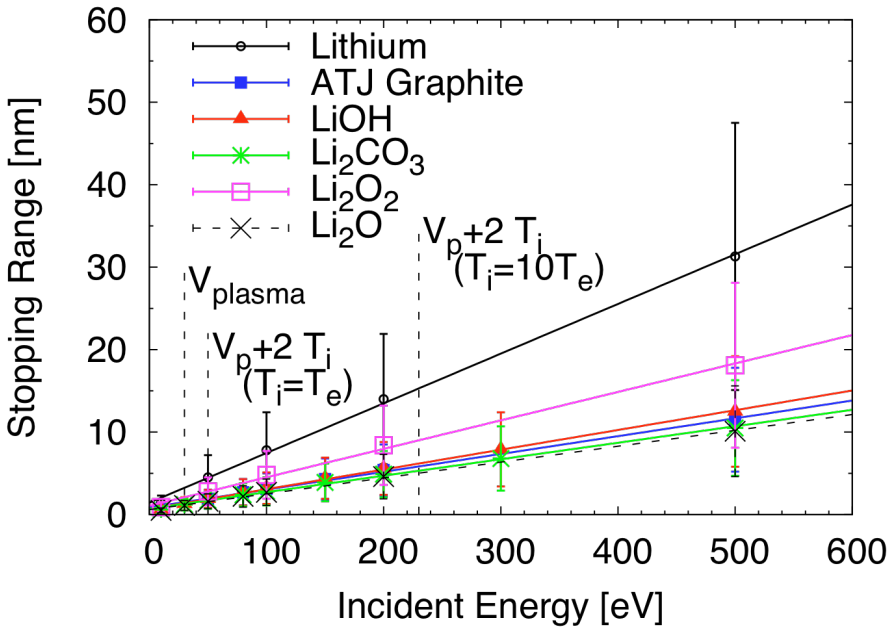
Lithium
ATJ Graphite
LiOH
Li2CO3
Li2O2
Li2O
Vplasma
Vp+2 Ti (Ti=Te)
Vp+2 Ti (Ti=10Te)
Stopping Range [nm]
Incident Energy [eV]

Fig. 8